October 30, 1986

Docket No.: 50-321

Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Exemption from certain containment leak testing requirements of Appendix J to 10 CFR Part 50 in response to your letter of March 5, 1979 as supplemented May 14, 1986. It has also issued the enclosed Amendment No. 131 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit No. 1. This Amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 5, 1979, as supplemented February 7, 1984. The Amendment revises the Hatch Unit No. 1 TSs to delete reference to specific dated versions of Appendix J. For reasons noted in the Safety Evaluation (SE) we did not approve the requested deletion of the TS tables listing containment penetrations and containment isolation valves. As indicated in the enclosed Notice of Denial, you may request a hearing on this matter.

A copy of the SE supporting the Exemption and the Amendment is enclosed. As noted in the SE, we also conclude that 1) the piping modifications proposed in your March 5, 1979 letter are acceptable with regard to proper testing of valves per the requirements of Appendix J; and 2) the updated containment leak rate test program submitted by your letter of March 5, 1979, is acceptable.

The Exemption is being forwarded to the Office of the Federal Register for publication. The Notice of Issuance of the Amendment will be included in the Commission's Bi-Weekly Notice.

Also enclosed for your information is a copy of an Environmental Assessment and Finding of No Significant Impact which has been published in the Federal Register.

Sincerely,

George Rivenbark, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Exemption
2. Amendment No. 131
3. Safety Evaluation w/attached
   Technical Evaluation Report
4. Environmental Assessment
5. Notice of Denial

cc w/enclosures:
See next page

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SNorris:nc	GRivenbark	JPulsipher	JKudrick	GLainas		DMuller
6/16/86	6/16/86	6/27/86	7/07/86	7/10/86	7/16/86	10/6/86

BWR:DBL:DD	BWR:DBL:D	*DBL.PSB	*Previously concurred
WHouston	RBernero	JHulman	
10/17/86	10/28/86	7/07/86	

Mr. J. T. Beckham, Jr.
Georgia Power Company

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. DPR-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated March 5, 1979 as supplemented February 7, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 30, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.7-5	3.7-5
3.7-6	3.7-6
3.7-6a	3.7-6a
3.7-15	3.7-15

4.7.A.2.b(2)

(c) The acceptance criteria for subsequent peak pressure tests shall require the new L_{am} not to exceed L_a.

(d) The allowable operational leak rate, L_{ao}, which shall be met prior to resumption of power operation following a test (either as measured or following repairs and retest) shall not exceed 0.75 L_a.

c. Corrective Action for Type A Tests

If leak repairs are necessary to meet the allowable operational leak rate, the integrated leak rate test need not be repeated provided local leakage measurements are conducted and the leak rate differences prior to and after repairs, when corrected to the test pressure and deducted from the integrated leak rate measurements, yield a leak rate value not in excess of the allowable operational leak rate.

d. Frequency for Type A Tests

After the initial preoperational leak rate test, two integrated leak rate tests shall be performed at approximately equal intervals between the major shutdowns for inservice inspection conducted at ten-year intervals. In addition, an integrated leak rate test shall be performed at the end of the ten-year interval, which may coincide with the inservice inspection shutdown period.

e. Type B Test - Leak Tests of Penetrations with Seals and Bellows (Tables 3.7-2 and 3.7-3)

Type B tests shall be performed under the program established in Appendix J of 10CFR Part 50.

4.7.A.2.e. Type B Test - Leak Tests of Penetrations with Seals and Bellows (Continued)(Tables 3.7-2 and 3.7-3)

(1) Primary containment components which seal or penetrate the pressure containing boundary of the containment shall be tested at a pressure not less than P_a. These components shall be tested at each major refueling shutdown or at intervals not to exceed two years.

(2) (a) The personnel air lock shall be tested at intervals not to exceed six months at P_a by pressurizing the compartment between the two air lock doors.

During intervals of door use when containment integrity is required, the door seals shall be tested at 10 psig after each opening.

(b) Personnel air lock leakage shall not exceed 0.05 L_a.

f. Type C Tests-Local Leak Tests of Containment Isolation Valves (Tables 3.7-1 and 3.7-4)
Type C tests shall be performed under the program established in Appendix J of 10CFR Part 50.

Containment isolation valves (except for main steam line isolation valves) shall be tested at a pressure not less than P_a. Type C tests shall be performed at each major refueling shutdown or at intervals not to exceed two years.

g. Acceptance Criteria for Type B and Type C Tests

The combined leakage rate of components subject to Type B and C tests shall be determined under the program established in Appendix J of 10CFR Part 50 and shall not exceed 0.6 L_a.

h. Main Steam Line Isolation Valves

The main steam line isolation valves shall be tested at a pressure of 1/2 P_a for leakage at least once per operating cycle. If a total leak rate of 11.5 scf per hour for any one main steam line isolation valve is exceeded, repairs and retest shall be performed to correct this condition.

4.7.E. References

2. "Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants", Topical Report BN-TOP-1, Revision 1, Bechtel Corp. Issued November 1, 1972.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING EXEMPTION FROM 10 CFR 50, APPENDIX J, AND

AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

1.0 INTRODUCTION

By letter dated August 7, 1985, the NRC requested Georgia Power Company (GPC) to review its containment leakage testing program for Edwin I. Hatch Nuclear Plant, Unit 1 (Hatch 1), and the associated Technical Specifications for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1973. Since by this date there were already many operating nuclear plants and a number more in advanced stages of design or construction, the NRC decided to have these plants reevaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirement of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. Subsequently, Section III.D.2 of Appendix J was revised effective October 22, 1980, and conformance is considered in our evaluation. These staff positions have since been applied in our review of the submittals filed by the licensee for Hatch 1. The results of our evaluation are provided below.

2.0 EVALUATION

Our consultant, the Franklin Research Center (FRC), has reviewed the licensee's submittals dated August 28, 1975, November 16, 1977, and March 5, 1979, and prepared the attached evaluation of containment leakage tests for Hatch 1. We have reviewed this evaluation and concur in its bases and findings, as modified below.

Several changes to the consultant's report should be noted.

A. The FRC identified six exemptions from the requirements of Appendix J; however, additional staff review has shown that four of the items are not bonafide exemptions, and another item, concerning air lock testing, is no longer an exemption because Appendix J has been revised and the proposed testing is now in compliance with Appendix J.

The following paragraphs discuss the six items which were identified as exemptions:

1. Isolation Valves Tested with Water

The licensee proposes to test certain isolation valves using water at a pressure of 1.10 Pa, in lieu of air, for systems which remain water-filled post LOCA. The measured leak rates are not included in the local leak rate test program result.

Appendix J to 10 CFR 50 requires that unless valves are sealed with fluid from a seal system, they shall be pressurized with air or nitrogen for leak testing purposes (Paragraph III.C.2). There are a number of valves, however, that are designed to remain covered with water after a LOCA and thus provide a water seal for the isolation valves or ensure that only liquid leakage from the containment will occur. For such valves, the licensee purposes to perform hydrostatic testing to determine their leak tightness. These valves fall into two categories, as discussed below.

A. Sealed by Water from the Torus

The following penetrations and systems are connected to the torus:

203	RCIC Pump Suction
204 A, B, C, D	RHR Pump Suction
207	HPCI Pump Suction
208 A, B	Core Spray Pump Suction
210 A, B	RHR/Core Spray Test Line

The piping for these systems penetrates the torus and terminates below the water line of the torus. As a supply of water in the torus is assured during post-accident conditions, these valves will remain sealed with water. Therefore, in accordance with Sections III.C.2 and III.C.3 of Appendix J, the valves need not be tested with air.

Although the licensee proposes to test them with water, this is not necessary, as the purpose of the water leak test is to assure a supply of sealing water for 30 days following onset of an accident. As the torus is postulated to always remain filled with water, no leak test is necessary to satisfy Appendix J requirements.

For the above reasons, the staff finds the proposed testing of the isolation valves in the above penetrations to be in compliance with the requirements of Appendix J.

B. Closed Systems Inside Containment

The following penetrations and systems are discussed in this section:

20	Service Water Supply
44	Service Water Return
23	Reactor Building Closed Cooling Water Supply
24	Reactor Building Closed Cooling Water Return

The Service Water and Reactor Building Closed Cooling Water (RBCCW) systems are closed systems inside containment. These closed systems constitute one of the two containment isolation barriers for each of the penetrations listed above, and are subject to ASME Section XI in-service inspection requirements. They are designed to remain intact and water filled post-LOCA. In accordance with Sections III.C.2 and III.C.3 of Appendix J, the licensee proposes to leak test the isolation valves in these systems with water at a pressure of 1.10 Pa; the leakage acceptance criteria are based upon maintaining a 30-day inventory of water for sealing the valves.

Therefore, the staff finds the proposed testing of the isolation valves in the above penetrations to be in compliance with the requirements of Appendix J.

2. Main Steam Isolation Valves

Appendix J to 10 CFR 50 requires leak rate testing of BWR main steam isolation valves (MSIVs) (Paragraph II.H.4) at Pa, the peak calculated containment pressure related to the design-basis accident (Paragraph III.C.2). Further, Appendix J requires that the measured leak rates be included in the summation for the local leak rate tests (Paragraph III.C.3).

The licensee proposes to leak test the MSIVs at a reduced pressure and exclude the measured leakage from the combined local leak rate test results. The staff has determined that an exemption to Appendix J is required for this proposal. The basis for this determination is discussed below.

Each main steam line is provided with two MSIVs that are oriented to seal in the direction of post-accident containment atmosphere out-leakage. The design of the MSIVs is such that testing in the reverse direction tends to unseat the valve. Simultaneous testing of the two valves, at design pressure, by pressurizing between the valves, would lift the disc of the inboard valve and result in a meaningless test. The proposed test calls for a test pressure of 28 psig (one-half of Pa) to avoid lifting the disc of the inboard valve. The total observed leakage through both valves (inboard and outboard) is then conservatively assigned to the penetration. The staff concludes that this procedure is acceptable. Furthermore, excluding the leakage from the summation for the local leak rate tests is acceptable because a separate leakage rate acceptance criterion of 11.5 standard cubic feet per hour is used for the MSIVs. This separate limit was found acceptable during the operating license review for Hatch 1, as discussed in Section 5.4.4 of the SER, dated May 11, 1973, and Supplement No. 1 to the SER, dated December 10, 1973. The radiological consequence of this separate leakage was considered generically as described by Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," Rev. 1, dated June 1976, which concluded that the Hatch 1 plant did not need to add such a leakage control system in order to reduce the radiological consequence. The separate limit of 11.5 scfh was also included in the original facility Technical Specifications.

The staff concludes that leak testing the MSIVs in the way described above is an acceptable alternative to the requirements of Appendix J, and that an exemption to Appendix J is justified and acceptable.

3. Air Lock

The licensee's proposal to test personnel air lock door seals by pressurizing the volume between the double door seals to a pressure of 10 psig required an exemption when originally proposed and reviewed by FRC. Because of a subsequent revision to Appendix J (October 22, 1980), this testing no longer requires an exemption, but rather complies with the current requirements of Appendix J.

4. Closed Systems Outside Containment

The following penetrations and systems are under discussion in this section:

12	RHR Suction
13 A, B	RHR Return to recirc.
16 A, B	Core Spray
17	RPV Head Spray
39 A, B	Containment Spray
211 A, B	Torus Spray

For each of these penetrations, the inner isolation barrier is an isolation valve that is Type C tested in accordance with Appendix J; the outer barrier is a closed system outside containment, having no containment isolation valve. Thus, the closed systems outside containment cannot be Type C tested. However, the systems are subject to the inservice inspection requirements of the AMSE Boiler and Pressure Vessel Code, Section XI, for Nuclear Class 2 piping, which requires that the entire closed system be pressurized and any visible leakage be repaired. They are visually inspected up to the containment isolation valves when the various system pump functional tests are performed. The leakage is not added to the local leak rate test program result for pneumatic testing. Because the isolation valves in these systems are Type C tested, the staff finds the testing program to be in compliance with the requirements of Appendix J.

5. Traversing Incore Probe System

The traversing incore probe system is equipped with a ball valve in each guide tube that provides shutoff capability following cable withdrawal. A shear valve is also provided for each guide tube to cut the cable and isolate the tube if the drive cable cannot be withdrawn.

The licensee will perform a Type C test on the ball valve. Because the shear valve requires testing to destruction, the licensee cannot perform periodic Type C tests on these valves. However, statistically chosen samples of the shear valves are tested by the manufacturer. Failure of a single shear valve to meet the 10^{-2} cc/sec leakage criterion results in rejection of the entire lot. The licensee has committed, by letter dated May 14, 1986, to test the explosive charges which operate the shear valves using procedures similar to those currently used for the standby liquid control system to comply with Hatch Unit 1 Technical Specification Sections 3.4/4.4. These Sections require that a portion of explosive charges installed in the valves be fired each refueling cycle to assure that the installed charges are operable and require that all installed charges are tested during the course of two fuel cycles. They also require that replacement charges be selected from batches that have been tested. Continuity of the electrical system that fires the charges to operate the shear valves is continuously monitored via indicating lights. Should continuity be lost, indicating light illuminate, and an alarm is received in the control room. Based on the above discussion, the staff concludes that the leak testing of the traversing incore probe system is acceptable, and no exemption is required, as the testable valves (ball valves) are Type C tested and the shear valves, which cannot be Type C tested, undergo alternative surveillance.

6. Control Rod Drive

The design of the Control Rod Drive (CRD) insert and withdraw lines does not facilitate Type C testing, as there are no containment isolation valves in these lines. However, adequate leakage monitoring of the CRD lines is provided by normal plant operating procedures and the Type A leakage rate tests. Since the insert and withdraw lines are pressurized to at least reactor operating pressure (1000 psi) by the cooling water flow during normal plant operation, leakage from these lines would be immediately evident.

The hydraulic control units are installed in a relatively high traffic area of the reactor building. In addition, plant procedure requires that an operator make a visual inspection of the CRD hydraulic control units (operating pressure 1000 psi) for leakage at least once per shift and that he record the inspection. Furthermore, because the reactor pressure vessel and nonseismic portion of the control rod drive system are vented during Type A tests, leakage monitoring of the control rod drive insert and withdraw lines is provided by Type A leakage rate tests.

The CRD system does not contain isolation valves that fall into the categories defined in Section II.H, "Type C tests," of Appendix J. Furthermore, because of the foregoing considerations, the CRD system does not constitute a potential containment atmosphere leak path. Therefore, the CRD system does not require Type C testing. The staff concludes that leakage monitoring of the control rod drive system in the manner described above meets the requirements of Appendix J.

B. Table 5 (page 8) of the report discusses a proposed change to Technical Specification 4.7.E to update the reference to Appendix J so as to include the latest revision of Appendix J. Subsequent to the writing of the consultant's report (April 1980), the licensee, by letter dated February 7, 1984, has determined that reference to a specific revision to Appendix J should be deleted from Technical Specification 4.7. This eliminates the need for future revisions to the Technical Specifications whenever Appendix J is revised. The Technical Specification is thereby, always consistent with the current Appendix J to 10 CFR 50. We conclude that this change is acceptable.

C. The FRC accepted the licensee's proposal to delete Tables 3.7-2, 3.7-3, and 3.7-4 from the current Technical Specifications. These tables contain list of primary containment penetration's with double O-ring seals, containment penetration's isolation valves, respectively. The licensee stated that, with respect to the updated program, these tables are inaccurate and incomplete. Rather than to include all of the penetrations and valves in these tables, the licensee determined that it would be more prudent to incorporate statements in the surveillance requirements outlining the programs for the Type A, B, and C tests to be in accordance with Appendix J. The tables would then be maintained as part of the plant's Appendix J program procedure.

However, based on additional review of this request the staff has determined that these tables should not be deleted from the Technical Specifications at this time, as they provide guidance to the NRC's regional inspectors in measuring the compliance of the licensee with the requirements of Appendix J. Deletion would also be contrary to current standard Technical Specifications. Therefore, the staff concludes that deletion of the tables is not acceptable at this time. However, as part of the staff's ongoing effort to generically improve Standard Technical Specifications, such a deletion may be reconsidered in the future. In the meantime, the licensee has informed the staff that updated tables will be submitted for inclusion in the Hatch 1 Technical Specifications.

Based on our review of the licensees request and on our review of the attached Technical Evaluation Report as prepared by the FRC, we have made the following conclusions regarding the Appendix J review for Hatch 1:

1. The updated containment leak rate test program submitted by GPC in the March 5, 1979, letter is acceptable. In addition, the associated proposed exemption from the requirements of Appendix J, concerning MSIV testing, is acceptable.

2. The proposed piping modifications submitted by GPC with its March 5, 1979, letter are acceptable with regard to the proper testing of valves per the requirements of Appendix J.

3. The proposed changes to the Technical Specifications requested by GPC in its March 5, 1979 letter, as supplemented by a February 7, 1984 letter, are acceptable, except as described in section C above.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment and Final Finding of No Significant Impact has been issued for the Exemption.

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10

CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this action will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Pulsipher

Dated: October 30, 1986

Attachment:
Technical Evaluation Report

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

NRC DOCKET NO. 50-231

NRC TAC NO. 10710

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 22

Prepared by

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Prepared for

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April 22, 1980

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The Benjamin Franklin Parkway, Phila., Pa 19103 (215) 448-1000

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE TESTING

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

CONTENTS

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE TESTING

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

1.0 BACKGROUND

On August 7, 1975 (1), the NRC requested the Georgia Power Company (GPC) to review the containment leakage testing program for Edwin I. Hatch Nuclear Plant Unit 1 (Hatch-1) and to provide a plan for achieving full compliance where necessary.

GPC responded on August 28, 1975 (2) by stating that the containment leak rate test program for Hatch-1 had been reviewed and the program was in full compliance with Appendix J. However, in a letter dated November 16, 1977 (3), GPC reported that in formulating a test program for Hatch-2, it discovered that the Hatch-1 program needed to be updated. Consequently, proposed changes to the Hatch-1 technical specifications were also submitted in the November 16, 1977 letter. In response to GPC's proposed changes, the NRC issued Amendment No. 53 to Facility Operating License No. DPR-57 for Hatch-1 on April 12, 1978 (4). In its letter of April 12, 1978 (4), the NRC indicated that Amendment No. 53 did not resolve all of GPC's proposed changes but that they would be reviewed as part of the review of the Hatch-2 program.

Subsequently, on March 5, 1979 (5), GPC submitted an updated containment leak rate test program which was developed utilizing the recently approved test program for Hatch-2. In addition to providing the updated program, the March 5, 1979 (5) letter also provided proposed changes to the technical specifications for Hatch-1 and proposed piping modifications, both of which were necessary for the full implementation of the updated program.

The purpose of this report is to provide a technical evaluation of the implementation of the requirements of 10CFR50, Appendix J, at Hatch-1. The submittal of March 5, 1979 (5) proposes a complete implementation package, including revised technical specifications, piping modifications, and an updated test program which supersedes previous correspondence for this topic. Consequently, this report addresses the technical acceptability of the proposed technical specification changes, proposed piping modifications, and the updated containment leak rate test program of the March 5, 1979 (5), submittal.

2.0 REVIEW CRITERION

10CFR50, Appendix J, Containment Leakage Testing was the NRC-provided criterion used to review GPC's submittal of March 5, 1979 (5).

3.0 TECHNICAL EVALUATION

GPC's letter of March 5, 1979 (5), provided:

1. Proposed technical specification changes necessary to implement an updated test program.
2. Proposed piping modifications necessary to perform testing of certain penetrations in accordance with the updated test program.
3. The updated containment leak rate test program itself.

Each of these items is evaluated in the remaining parts of this section.

3.1 PROPOSED TECHNICAL SPECIFICATION CHANGES

3.1.1 Proposal to Delete Tables Listing Primary Containment Penetrations with Double O-Ring Seals, Containment Penetrations with Expansion Bellows and Containment Isolation Valves

The proposed technical specification changes submitted by GPC, along with the updated containment leak rate test program, deleted Tables 3.7-2, 3.7-3, and 3.7-4 (constituting deletion of Pages 3.7-21 through 3.7-27) from the current technical specifications. These tables contained lists of primary containment penetrations with double O-ring seals, containment penetrations with expansion bellows, and containment isolation valves, respectively. GPC stated that with respect to the updated program, these tables were inaccurate and incomplete. Rather than to include all of the penetrations and valves in these tables, GPC determined that it would be more prudent to incorporate statements in the surveillance requirements outlining the programs for the Type A, B, and C tests to be in accordance with Appendix J. GPC's intent was to be more comprehensive than it would be if it were to attempt to list each penetration in the technical specifications as is done in the present leak rate test program. Additionally, GPC stated that it believed that such detailed information was not proper for inclusion in the technical specifications and that this approach was consistent with the Hatch-2 standard technical specifications on this subject.

Evaluation. GPC's proposed updated containment leak rate test program removes from the technical specifications the lists of penetrations and isolation valves which are periodically tested, while retaining these lists in the containment leak test program itself. At the same time, GPC proposes to modify the technical specifications to require that penetrations with O-ring seals or expansion bellows be tested in accordance with Type B procedures of Appendix J. This approach is not only acceptable but appears to be preferable to the current program at Hatch-1 because it will be consistent with the recently approved program at Hatch-2 and also because it will enable GPC to revise and update more readily the specific penetrations and valves listed in the containment leak rate test program without having to modify the technical specifications. At the same time, the technical specifications will clearly require conformance with Appendix J.

In view of the above considerations and the revisions of the technical specifications evaluated in Section 3.1.2, below, Franklin Research Center (FRC) finds the proposal to delete Tables 3.7-2, 3.7-3, and 3.7-4 (deletion of Pages 3.7-21 through 3.7-27) from the current technical specifications to be acceptable.

3.1.2 Proposal to Revise Specifications for Type B Testing of Penetrations with Seals and Bellows, Type C Testing of Isolation Valves, Acceptance Criteria for Type B and Type C Tests, Main Steam Isolation Valves and References

In order to support its updated containment leak rate test program, GPC proposes to revise Pages 3.7-5, 3.7-6, and 3.7-6a of the current technical specifications.

GPC's basis for revising these pages is to provide a program consistent with the containment leak rate test program for Hatch-2 and to require compliance with Appendix J. An evaluation of each of these proposed changes is provided in Tables 1 through 5 (Pages 4 through 8).

TABLE 1

Specification 4.7.A.2.e

TITLE	PRESENT WORDING	PROPOSED WORDING
Type B Test - Leak Tests of Penetrations with Seals and Bellows	(1) Primary containment components which seal or penetrate the pressure boundary of the containment shall be tested at a pressure not less than Pa. These components shall be tested at each major refueling shutdown or at intervals not to exceed two years.	Type B tests shall be performed under the program established in Appendix J of 10CFR Part 50 (Reference 1). (1) Primary containment components which seal or penetrate the pressure containing boundary of the containment shall be tested at each major refueling shutdown or at intervals not to exceed two years.

EVALUATION

Since the proposed change has no effect on the current specification other than to require that the Type B tests be performed under the program established in Appendix J, this proposed specification change is acceptable.

TABLE 2

Specification 4.7.A.2.f

TITLE	PRESENT WORDING	PROPOSED WORDING
Type C Tests - Local Leak Tests of Containment Isolation Valves	Containment isolation valves (except for main steam line isolation valves) shall be tested at a pressure not less than Pa. Type C tests shall be performed at each major refueling shutdown or at intervals not to exceed two years.	Type C tests shall be performed under the program established in Appendix J of 10CFR Part 50 (Reference 1). Containment isolation valves (except for main steam isolation valves) shall be tested at a pressure not less than Pa. Type C tests shall be performed at each major refueling shutdown or at intervals not to exceed two years.

EVALUATION

Since the proposed change has no effect on the current specification other than to require that the Type C tests be performed under the program established in Appendix J, this proposed specification change is acceptable.

TABLE 3

Specification 4.7.A.2.g

TITLE	PRESENT WORDING	PROPOSED WORDING
Acceptance Criteria for Type B and Type C Tests	The combined leakage rate of components subject to Type B and C tests (except for main steam line isolation valves) shall not exceed 0.6 La.	The combined leakage rate of components subject to Type B and C tests shall be determined under the program established in Appendix J of 10CFR Part 50 (Reference 1) and shall not exceed 0.6 La.

EVALUATION

Since the proposed change has no effect on the current specification other than to require that the combined leakage rate of the components subject to Type B and C tests be determined under the program established in Appendix J, this proposed specification change is acceptable.

TABLE 4

Specification 4.7.A.2.h

TITLE	PRESENT WORDING	PROPOSED WORDING
Main Steam Line Isolation Valves	The main steam line isolation valves shall be tested at a pressure of 28 psig for leakage at least once per operating cycle. If a total leak rate of 11.5 scf per hour for any one main steam line isolation valve is exceeded, repairs and retest shall be performed to correct this situation.	The main steam line isolation valves shall be tested at a pressure of 1/2 Pa for leakage at least once per operating cycle. If a total leak rate of 11.5 scf per hour for any one main steam line isolation valve is exceeded, repairs and retest shall be performed to correct this condition.

EVALUATION

At Hatch-1, 28 psig equals 1/2 Pa therefore the proposed change is technically identical to the present specification. By revising the specification to replace the value of psig with 1/2 Pa, this specification becomes consistent with the rest of the technical specifications which require testing in terms of Pa rather than a specific psig value, and it also is preferable since the specification remains valid regardless of any future changes or revisions to the analytically determined value of Pa. Consequently, this proposed change to the specification is evaluated as being acceptable.

TABLE 5

Specification 4.7.E

TITLE	PRESENT WORDING	PROPOSED WORDING
References	Reactor Containment Leakage Testing for Water Cooled Power Reactors, Appendix J to 10CFR50.54 (o) February 14, 1973.	Reactor Containment Leakage Testing for Water Cooled Power Reactors, Appendix J to 10CFR50.54 (o) February 14, 1973 as corrected and amended through April 19, 1976.

EVALUATION

Since this proposed change updates the current specification to reflect the latest amendment of Appendix J, the proposed specification change is acceptable.

3.2 PROPOSED PLANT MODIFICATIONS

In its letter of March 5, 1979 (5), GPC submitted the description of several plant modifications needed for various penetrations to facilitate leak rate testing in accordance with Appendix J and the updated test program for Hatch-1. GPC requested that the NRC approve these proposed modifications prior to GPC's initiation of procurement and engineering design activities. A technical evaluation of each of the proposed modifications is provided in Table 6 of this report (pages 10 and 11).

As shown in Table 6, all proposed piping modifications are considered to be acceptable with regard to proper testing of valves per Appendix J requirements. Initiation of activities to ensure installation of these modifications as soon as practical is considered to be essential since full implementation of the updated containment leak rate test program at Hatch-1 is contingent upon successful completion of the modifications.

3.3 THE UPDATED CONTAINMENT LEAK RATE TEST PROGRAM

The updated program provides a penetration leakage rate test list which describes the inboard and outboard isolation barriers for each primary containment penetration at Hatch-1. GPC has compiled the list in tabular form; it includes the type of test required for each penetration barrier, certain special notes where applicable, and references to drawing numbers. The list was prepared with the assumption that the proposed technical specification changes of Section 3.1, above, would be approved and incorporated and that the proposed piping modifications of Section 3.2 above, would be accomplished.

GPC stated that the basis used to establish the testing requirements and acceptance criteria for the Hatch-1 program was identical to that used, and recently approved by the NRC, for Hatch-2. In particular, GPC compared each penetration of Hatch-1 to its similar penetration of Hatch-2 and evaluated it under the same guidelines used to develop the Hatch-2 program. In addition to providing a program for Hatch-1, which is based upon a program previously approved by the NRC, the updated program was intended by GPC to provide continuity of testing procedures between the two Hatch units.

Evaluation. Since GPC developed this updated test program by comparing each penetration at Hatch-1 with its similar penetration at Hatch-2 and applying the same guidelines used to develop the Hatch-2 program, the updated test program

TABLE 6

Evaluation of Proposed Piping Modifications

PENETRATION	PURPOSE	DESCRIPTION	EVALUATION
X-25 - Vent Purge Supply	To allow testing of Valve T48-F118A in the proper direction.	Addition of one 2" and one 3/4" ASME Section III, Class 2 valve between T48-F118A and Penetration X-25.	This modification will achieve its purpose and is considered acceptable.
X-26 - Vent Purge Return/H_2 & O_2 Analyzer	To allow testing of Valves T48-F335A and B, and P33-F002 in the proper direction.	Addition of two 3/4" and one 2" ASME Section III, Class 2 valves. One 3/4" valve added as a test fitting between Valves P33-F002 and P33-F119 and the other 3/4" plus the 2" valves added to line 2" HMD between Valves T48-F335A and B and Penetration X-26.	This modification will achieve its purpose and is considered acceptable.
X-28A - Recirculation Sample	To allow testing of Valve B31-F019 in the proper direction.	The addition of a test fitting containing two 3/4" ASME Section III, Class 1 valves in series between Valves B31-F019 and B31-F059.	This modification will achieve its purpose and is considered an acceptable configuration.
X-28F - H_2 & O_2 Analyzer	To allow testing of Valve P33-F003.	The addition of a test fitting with one 3/4" ASME Section III, Class 2 valve between Valves P33-F003 and P33-F120.	This modification will achieve its purpose and is considered an acceptable configuration.
X-31F - Recirculation Pump Seal Water	To provide testing capabilities for Check Valves B31-F013A and B31-F017A.	The addition of one test fitting upstream of each check valve, each test fitting containing two ASME Section III, Class 2 valves (3/4").	This modification will achieve its purpose and is considered an acceptable configuration.
X45F - ILRT Verification Flow	To allow testing of Valve T23-F004 in the proper direction.	Installing a flange on the pipe termination inside the drywell and testing through a blind flange with an installed test connection.	This modification will achieve its purpose. Since the blind flange will have an installed test fitting, there is little chance that the blank will be left installed after the test since it will be removed with the test equipment. Therefore, this modification is considered

TABLE 6 (continued)

PENETRATION	PURPOSE	DESCRIPTION	EVALUATION
X46 - Demineralized Water	To allow testing of Valves P21-F406 and P21-F353 in the proper direction.	Relocating of Valves P21-F372, P21-F406, and P21-F353 so that P21-F353 can be tested through test fitting P21-F407 and so that Valve P21-F406 can be tested by pressurizing through a drywell hose connection.	This modification will achieve its purpose. Since no additional valves are required, but the position of the valves is merely changed, this configuration is considered acceptable.
X59A - Recirculation Pump Seal Water	To allow testing of Check Valves B31-F013B and B31-F017B.	Same as Penetration X31F	This modification is identical to Penetration X31F and is therefore also acceptable.
X205 - Containment Purge and Inerting	To allow testing of Valve T48-F118B in the proper direction.	Addition of one 2" and one 3/4" ASME Section III, Class 2 valve between T48-F118B and Penetration X205.	This modification is identical to Penetration X25 and is therefore also acceptable.
X210 - Radwaste Connection	To remove the radwaste connection to the Core Spray System since the tie-in is not required and the Quality Group D radwaste tie-in prevents the Core Spray System from being considered a closed system.	The G11 tie-in to Core Spray Line 10" HLB is to be removed and capped.	Removal of the Quality Group D tie-in to the Quality Group B Core Spray System is considered an acceptable method to restore the Quality Group B integrity of the design of the Core Spray System.
S217 - H_2 & O_2 Analyzer	To allow testing of Valve P33-F007 in the proper direction.	Addition of a test fitting with one 3/4" ASME Section III, Class 2 valve between valves P33-F007 and P33-F126.	This modification will achieve its purpose and is considered an acceptable configuration.
X220 - Vent Purge Outlet/H_2 & O_2 Analyzer	To allow testing of Valves T48-F333A and B, and P33-F006 in the proper direction.	Addition of two 3/4" and one 2" ASME Section III, Class 2 valves. The configuration is the same as that described in Penetration X26, above.	This modification is identical to Penetration X-26 and is therefore also acceptable.

at Hatch-1 should meet all the requirements of 10CFR50, Appendix J. To ensure that the guidelines utilized in the development of the Hatch-2 program were properly carried over and applied to Hatch-1, the updated Hatch-1 program was independently reviewed in detail by FRC as part of this evaluation.

This review by FRC revealed that the program contains certain exemptions from the requirements of Appendix J which are acceptable from the standpoint of technical equivalence to the requirements or that there is sufficient basis for the exemption to ensure that the intent of Appendix J is satisfied. These acceptable exemptions include:

- Testing of isolation valves with water at a pressure of 1.10 Pa in lieu of air in systems which remain water-filled post-LOCA. Leakage is not included in the 0.6 La total.
- Testing of Main Steam Isolation Valves at 1/2 Pa with an acceptance criteria of 11.5 scfh for any valve. Leakage is not included in the 0.6 La total.
- Testing of personnel airlock seals between the double seals at a pressure of 10 psig.
- Utilizing closed systems outside containment as an isolation barrier where the system is subject to the in-service inspection requirements of ASME, Section XI, for Nuclear Class 2 piping and the system remains filled with water and operating at a pressure greater than Pa post-LOCA. The in-service inspection requires that any visible leakage be repaired. The leakage results are not added to the 0.6 La total.
- Excluding traveling in-core probe (TIP) drive shear valves from Type C testing because the shear valves are designed to destroy the tubes when required to function. However, the inboard TIP ball valves are Type C tested and also, each lot of shear valves are sample leakage tested by the manufacturer prior to delivery. Failure of a single shear valve to meet the 10^{-2} cc/sec leakage criteria results in rejection of the entire lot. Explosive charges, which operate the shear valves, are in-service inspected in accordance with the requirements of ASME, Section XI.
- Testing of control rod drive (CRD) insert and withdraw lines by Type A procedures but not Type C procedures. These lines are continuously monitored for leakage (at least every four hours) during reactor operation at pressures which are at least equivalent to reactor operating pressure.

In light of the prior review and acceptance of these exemptions by the NRC during the review of the Hatch-2 program and the independent review conducted by FRC, the updated containment leak test program for Hatch-1 is considered to be acceptable from the standpoint of satisfying the intent of Appendix J.

4.0 CONCLUSIONS

As a result of the technical evaluation provided in Section 3.0, above, FRC concludes that:

- The proposed technical specification changes submitted by GPC with its March 5, 1979 (5), letter are technically acceptable.
- The proposed piping modifications submitted by GPC with its March 5, 1979 (5), letter are acceptable with regard to proper testing of valves per Appendix J requirements and should be implemented as soon as practical in order to support the updated containment leak rate test program.
- The updated containment leak rate test program submitted by GPC in the March 5, 1979 (5), letter is technically acceptable.

5.0 REFERENCES

1. NRC generic letter to GPC, dated August 7, 1975, concerning the implementation of 10CFR50, Appendix J, at Hatch-1.
2. GPC letter from Mr. C. R. Whitmer (Vice President, Engineering) to Mr. A. Giambusso (Director, Division of Operating Reactors, USNRC) dated August 28, 1975.
3. GPC letter from Mr. C. F. Whitmer (Vice President, Engineering) to Mr. V. Stello, Jr. (Director, Division of Operating Reactors, USNRC) dated November 16, 1977.
4. NRC letter from Mr. V. Stello, Jr. (Director, Division of Operating Reactors) to Mr. C. F. Whitmer (Vice President, Engineering, GPC) dated April 14, 1978.
5. GPC letter from Mr. C. F. Whitmer (Vice President, Engineering) to Director of Nuclear Reactor Regulation, USNRC, dated March 5, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

GEORGIA POWER COMPANY

DOCKET NO. 50-321

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from the requirements of Appendix J to 10 CFR 50 to Georgia Power Company (GPC), Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, the licensees for the Edwin I. Hatch Nuclear Plant, Unit No. 1 (Hatch 1), located in Appling County, Georgia.

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action: In accordance with GPC's request dated March 5, 1979, the exemption would permit the licensees to leak test the Main Steam Isolation Valves at 28 psig with an acceptance criteria of 11.5 scfh for any valve. Leakage from these valves will not be included in the summation of the local leak rates.

The Need for the Proposed Action: 10 CFR 50.54(o) requires that primary reactor containments for water cooled power reactors be subject to the requirements of Appendix J to 10 CFR Part 50. Appendix J contains the leakage test requirements, schedules, and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. Appendix J was published on February 14, 1973, and by letter dated August 7, 1975, the Commission requested GPC to review the containment leakage testing program for the facility and to provide a plan for achieving full compliance where necessary.

GPC responded on August 28, 1975, by stating that the containment leak rate test program for Hatch 1 had been reviewed and the program was in full compliance with Appendix J. However, in a letter dated November 16, 1977, GPC reported that in formulating a test program for the Edwin I. Hatch Nuclear Plant, Unit 2 (Hatch 2), it discovered that the Hatch 1 program needed to be updated. Consequently, proposed changes to the Hatch 1 Technical Specifications were also submitted in the November 16, 1977 letter. In response to GPC's proposed changes, the Commission issued Amendment No. 53 to Facility Operating License No. DPR-57 for Hatch 1 on April 12, 1978. In its letter of April 12, 1978, the Commission indicated that Amendment No. 53 did not resolve all of GPC's proposed changes but that they would be reviewed as part of the review of the Hatch 2 program.

Subsequently, on March 5, 1979, GPC submitted an updated containment leak rate test program which was developed utilizing the then recently-approved test program for Hatch 2. In its review of this March 5, 1979 submittal, the staff determined that an exemption to the requirement of 10 CFR 50, Appendix J is required for the proposed testing of the main steam isolation valves (MSIVs) so that they may be tested at 1/2 the Appendix J required pressure and so that the leakage through the MSIV's is not required to be added in the summation of the leakage from the other isolation valves and penetrations.

Environmental Impact of the Proposed Action: The proposed exemption to the Appendix J test requirements for the MSIV's will not cause post-accident radiological releases to exceed those determined previously for Hatch 1. The proposed exemption does not otherwise affect facility radiological effluents, or any significant occupational exposures. Likewise, the proposed exemption does not affect facility nonradiological effluents and has no other environmental impact. Therefore, the Commission concludes there are no measurable radiological or nonradiological environmental impacts associated with the proposed exemption.

Since the Commission has concluded there is no measurable environmental impact associated with the proposed exemption, any alternatives either will have no environmental impact or will have a greater environmental impact. The principal alternative to the exemption would be to require literal compliance with Appendix J to 10 CFR Part 50. Such an action would not enhance the protection of the environment.

Alternative Use of Resources: This action does not involve the use of resources not considered previously in connection with the Final Environmental Statement (FES) relating to this facility, FES for Edwin I. Hatch Units 1 and 2, USAEC (October 1972).

Agencies and Persons Consulted: The Commission's staff reviewed GPC's request and did not consult other agencies or persons.

FINDING OF NO SIGNIFICANT IMPACT

The Commission has determined not to prepare an environmental impact statement for the proposed exemption.

Based upon the environmental assessment, we conclude that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the request for exemption dated March 5, 1979, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

Dated at Bethesda, Maryland, this 7th day of October 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
GEORGIA POWER COMPANY, ET AL	)	Docket No. 50-321
	)	
(Edwin I. Hatch Nuclear Plant, Unit No. 1)	)	

EXEMPTION

I.

The Georgia Power Company (GPC or the licensee) and three other co-owners are the holders of Facility Operating License No. DPR-57 which authorizes operation of the Edwin I. Hatch Nuclear Plant, Unit 1 (Hatch 1 or the facility) at steady state reactor power levels not in excess of 2436 megawatts thermal. The facility is a boiling water reactor located at the licensee's site in Appling County, Georgia. The license is subject to all rules and regulations and orders of the Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

II.

Section 50.54(b) of 10 CFR 50 requires that primary reactor containments for water cooled power reactors be subject to the requirements of Appendix J to 10 CFR Part 50. Appendix J contains the leakage test requirements, schedules, and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. Appendix J was published on February 14, 1973, and by letter dated August 7, 1975, the Commission requested GPC to review the containment leakage testing program for the facility and to provide a plan for achieving full compliance where necessary.

GPC responded on August 28, 1975, by stating that the containment leak rate test program for Hatch 1 had been reviewed and the program was in full compliance with Appendix J. However, in a letter dated November 16, 1977, GPC reported that in formulating a test program for the Edwin I. Hatch Nuclear Plant, Unit 2 (Hatch 2) it discovered that the Hatch 1 program needed to be updated. Consequently, proposed changes to the Hatch 1 Technical Specifications were also submitted in the November 16, 1977 letter. In response to GPC's proposed changes, the Commission issued Amendment No. 53 to Facility Operating License No. DPR-57 for Hatch 1 on April 12, 1978. In its letter of April 12, 1978, the Commission indicated that Amendment No. 53 did not resolve all of GPC's proposed changes but that they would be reviewed as part of the review of the Hatch 2 program.

Subsequently, on March 5, 1979, GPC submitted an updated containment leak rate test program which was developed utilizing the recently-approved test program for Hatch 2. In addition to providing the updated program, the March 5, 1979, letter also provided proposed changes to the Technical Specifications for Hatch 1 and proposed piping modifications, both of which were necessary for the full implementation of the updated program.

Since GPC developed this updated test program by comparing each penetration at Hatch 1 with its similar penetration at Hatch 2 and applying the same guidelines used to develop the Hatch 2 program, the updated test program at Hatch 1 should meet all the requirements of 10 CFR 50, Appendix J. To ensure that the guidelines utilized in the development of the Hatch 2 program were properly carried over and applied to Hatch 1, the updated Hatch 1 program was independently reviewed in detail by our contractor, the Franklin Research Center (FRC). FRC prepared a Technical Evaluation Report (TER) "Containment Leakage Rate-Testing - Edwin I Hatch Nuclear Plant Unit 1" dated April 22, 1982, documenting the results of its review of GPC's March 5, 1979 submittal.

The TER identified six proposed test items as exceptions to the requirements of Appendix J and determined that exemptions to the requirements of Appendix J were required as to these six items. These items concern: 1) isolation valves tested with water 2) main steam isolation valves (MSIVs) 3) airlocks 4) closed systems outside containment 5) transversing incore probe system and 6) control rod drive lines. However, additional staff review, documented in the Safety Evaluation Report (SER), has shown that only the MSIV test item is an exception to the Appendix J requirements and that the other five items are in compliance with Appendix J. This additional staff review included consideration of additional information concerning items 4 and 5 above that was provided by the licensee in a May 14, 1986 submittal.

III

Appendix J to 10 CFR 50 requires leak rate testing of BWR main steam isolation valves (MSIVs) (Paragraph II.H.4) at Pa, the peak calculated containment pressure related to the design-basis accident (Paragraph III.C.2). Further, Appendix J requires that the measured leak rates be included in the summation of the leak rates for the local leak rate tests of all penetrations and valves subject to Type B and C tests (Paragraph III.C.3).

The licensee proposes to leak test the MSIVs at a reduced pressure and exclude the measured leakage from the combined local leak rate test results.

Each main steam line is provided with two MSIVs that are oriented to seal in the direction of post-accident containment atmosphere out-leakage.
The design of the MSIVs is such that testing in the reverse direction tends to unseat the valve. Simultaneous testing of the two valves, at design pressure,

by pressurizing between the valves, would lift the disc of the inboard valve and result in a meaningless test. The proposed test calls for a test pressure of 28 psig (one-half of Pa) to avoid lifting the disc of the inboard valve. The total observed leakage through both valves (inboard and outboard) is then conservatively assigned to the penetration. The staff concludes that this procedure is acceptable based on the conservative test direction for the inboard valve. Furthermore, excluding the leakage from the summation for the local leak rate tests is acceptable because a separate leakage rate acceptance criterion of 11.5 standard cubic feet per hour is used for the MSIVs. The separate limit of 11.5 scfh was also included in the original facility Technical Specifications. This separate limit was found acceptable during the operating license review for Hatch 1, as discussed in Section 5.4.4 of the SER, dated May 11, 1973, and Supplement No. 1 to the SER, dated December 10, 1973. The radiological consequence of this separate leakage was considered generically as described by Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," Rev. 1, dated June 1976, which recommended the installation of a supplemental control system for plants with construction permits issued after March 1, 1970, but concluded that the Hatch 1 plant and other plants for which construction permits were issued prior to March 1, 1970 did not need to add such a leakage control system.

Pursuant to Final Rule 10 CFR 50.12 (50 FR 50764) published on December 12, 1985, the special circumstances for granting this exemption have been identified, as follows. The purpose of the requirements to leak test the MSIVs at Pa is to assure that pressure conditions during testing represent pressure conditions that could be experienced in a design-basis accident so that potential leakage during a design-basis accident will be identified adequately during testing. However, as noted above, application of this requirement to valves with configurations similar to these MSIVs tends to unseat the valves and give meaningless results and would not serve the underlying purpose of the rule. The proposed alternate test, while at a somewhat reduced pressure, conservatively treats the resulting leakage indication and provides a more meaningful indication of potential leakage across the valves. Accordingly, with respect to the exemption from the requirement for full pressure testing, application of the rule in this instance would not serve the underlying purpose of the rule.

The purpose of the requirement to include the measured leak rates of the MSIVs in the summation of the local leak rate tests for all of the penetrations and valves subject to Type B and C tests is to assure that there is adequate margin between the detected combined valve leakage and the leakage limit. Experience has demonstrated that adequate margin can be maintained even if leakage from MSIVs is considered separately and subject to a separate specific leakage restriction of 11.5 standard cubic feet per hour. Accordingly, with respect to the exemption from the requirement to combine the result of all valve leakage tests, application of the rule in this instance is not necessary to achieve the underlying purpose of the rule. Consequently, special circumstances described by 10 CFR 50.12(a)(2)(ii) exist in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule in that the licensee has proposed an acceptable alternative test method that accomplishes the intent of the regulation.

The staff concludes that leak testing the MSIVs in the way described above is an acceptable alternative to the requirements of Appendix J, and that an exemption to Appendix J is justified and acceptable.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; furthermore, in accordance with 10 CFR 50.12(a)(ii) special circumstances, as discussed above, are present. Therefore, the Commission hereby grants the exemption identified above.

Pursuant to 10 CFR 51.32, the Commission has determined that the issuance of the exemption will have no significant impact on the environment (51 FR 36762).

A copy of the Commission's concurrently issued Safety Evaluation related to this action is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

This Exemption is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Bernero

Robert M. Bernero, Director
Division of BWR Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 30th day of October 1986

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

GEORGIA POWER COMPANY, ET AL.

DOCKET NO. 50-321

NOTICE OF DENIAL OF AMENDMENT TO

FACILITY OPERATING LICENSE AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied in part a request by the licensee for an amendment to Facility Operating License No. DPR-57, issued to the the Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia (the licensee), for operation of the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility), located in Appling County, Georgia.

The amendment, as proposed by the licensee, modified the Hatch Unit 1 Technical Specifications, Section 3.7 to delete Tables 3.7-2, 3.7-3 and 3.7-4. These Tables contain lists of primary containment penetrations and containment isolation valves. It also deletes the reference to a specific revision of Appendix J. The licensee's application for the amendment was dated March 5, 1979 and supplemented February 7, 1984. Notice of Consideration of Issuance of this amendment was published in the FEDERAL REGISTER on April 25, 1984 (49 FR 17860). All of the requested changes were granted, except the request to delete Tables 3.7-2, 3.7-3 and 3.7-4.

Notice of Issuance of Amendment No.131 will be published in the Commission's Bi-Weekly FEDERAL REGISTER Notice.

The portion of the application which requested deletion of Tables 3.7-2, 3.7-3 and 3.7-4 was denied.

The request to delete Tables 3.7-2, 3.7-3 and 3.7-4 was found to be unacceptable because they provide guidance in measuring the compliance of the licensee with the requirements of Appendix J and because deletion would be contrary to current Standard Technical Specifications.

The licensee was notified of the Commission's denial of this request by letter dated

By the licensee may demand a hearing with respect to the denial described above and any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for a hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H. Street, N.W., Washington, D.C., by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20036.

For further details with respect to this action, see (1) the application for amendment dated March 5, 1979, as supplemented February 7, 1984, and (2) the Commission's letter to Georgia Power Company dated October 30, 1986 which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georiga. A copy of Item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of BWR Licensing.

Dated at Bethesda, Maryland, this 30th day of October 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing